March 12, 2007

L-MT-07-012
10 CFR Part 50.73

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Monticello Nuclear Generating Plant
Docket No. 50-263
License No. DPR-22

LER 2007-001, "Reactor Scram due to Turbine Control Valve Housing Support Failure"

A Licensee Event Report for this occurrence is attached.

This letter contains no new commitments and no revisions to existing commitments.

John T. Conway

John T. Conway
Site Vice President, Monticello Nuclear Generating Plant
Nuclear Management Company, LLC

Enclosure

cc: Administrator, Region III, USNRC
Project Manager, Monticello, USNRC
Resident Inspector, Monticello, USNRC

IE22

NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 6-30-2007
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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TITLE (4) Reactor Scram due to Turbine Control Valve Housing Support Failure

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
01	10	2007	2007	– 001	– 00	03	12	2007	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	1
POWER LEVEL (10)	088

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply) (11)

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)	X	50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Ron Baumer	763-295-1357

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT

On January 10, 2007, the turbine control valves [JJ] ramped from approximately 50 percent open to 100 percent open in two seconds without a demand from the pressure control system. A Group 1 isolation and automatic scram ensued. Review of plant conditions immediately after the scram disclosed that the main turbine [TRB] control valves [SCV] did not close following the scram. During a post scram plant walk down it was identified that the supports for the turbine control valve enclosure [SPT] had failed, allowing the enclosure to transition downward approximately six inches.

The root cause of this event was determined to be latent shortcomings in the design and construction of the turbine control valve enclosure support system during initial construction. Corrective actions include modifying and strengthening the enclosure support system, performing inspections of associated piping and systems, and performing an extent of condition review.

LICENSEE EVENT REPORT (LER)

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

Description

At approximately 1528 on January 10, 2007, with the unit stable in Mode 1 at approximately 88 percent power, all four turbine control valves [SCV] ramped from approximately 50 percent open to 100 percent open in approximately two seconds with no demand to do so from the pressure control system. The resultant decrease in reactor pressure resulted in a group 1 isolation on low steam line pressure with the MODE switch in RUN, and associated reactor scram. The closure of the Main Steam [SB] Isolation Valves and reactor scram caused reactor pressure to increase and reactor water level to decrease to minus eight inches. The low reactor water level also resulted in a Group II Emergency Safety Feature actuation as designed. All control rods [AA] inserted and all safety systems responded as expected. During the reactor water level transient, the turbine [JJ] and main feed [SJ] pumps [P] tripped at 48 inches as designed. Maximum water level reached was 78 inches (30 inches below the main steam lines). Control of plant pressure was initially established by manually operating the safety relief valves (SRVs) [RV]. At no point was automatic operation of the SRVs required or challenged. Subsequently, reactor pressure control was established using the high pressure coolant injection system (HPCI) [BJ]. Reactor makeup was established using normal feedwater.

Post scram review disclosed that the turbine control valves did not close on the turbine trip as designed. A post scram walk down revealed that supports [SPT] for the control valve enclosure had failed, allowing the enclosure to transition downward approximately six inches, causing deformation of adjoining support structures. No system breach occurred and there was no release of radioactivity.

Following plant stabilization, the decision was made to proceed to cold shutdown (Mode 4). As uncertainty existed regarding the status of the main steam system (the control valve enclosure contacted insulation on a section of the main steam line and the control valves failed to close following the turbine trip), the decision was made to cool down using HPCI and the SRVs for pressure control, main feed for makeup, and the residual heat removal system [BO]. Mode 4 was achieved at 0613 on January 11, 2007.

Event Analysis

Pursuant to 10CFR 50.72 paragraphs (b)(2)(iv)(B) for the RPS actuation and paragraph (b)(3)(iv)(A) as an ESF actuation, an eight-hour event notification was made to the USNRC. Per 10 CFR 50.73 (a)(2)(iv), a Licensee Event report is required for this event.

The event is not classified as a safety system functional failure.

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TEXT *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

Safety Significance

At approximately 88 percent reactor power, all turbine control valves ramped full open. As reactor pressure dropped, reactor power lowered. The Group 1 isolation occurred at 840 psi as required. In response to the automatic scram at a MSIV position of 90 percent open with the MODE switch in RUN, all control rods inserted as designed. The minimum reactor water level reached during the transient was minus eight inches. No Emergency Core Cooling System (ECCS) actuation was required or occurred. As level lowered below nine inches, the Group 2 isolation occurred as expected. With the feed system in service, feedwater, in conjunction with decay heat, caused reactor water level to rise. At 48 inches the main turbine and the main feedwater pumps tripped as expected. The highest water level reached was 78 inches, approximately 30 inches below the main steam lines. Reactor pressure was controlled initially by manually operating the safety relief valves. No automatic operation of the safety relief valves was required. Water level was subsequently restored to normal levels, feedwater was restarted for makeup, and high pressure coolant injection started and operated in the pressure control mode to remove decay heat.

With the exception of the failure of the turbine control valves to close, all systems and components responded as expected to mitigate the transient and assure core cooling. Automatic operation of the SRVs and emergency core cooling systems did not occur and were not required. No safety systems or systems credited in the Probabilistic Risk Assessment (PRA) as capable of mitigating an accident were compromised as a result of the structural failure. As a result, there was no significant increase in the risk of a core damage accident or a release of radioactive material to the environment. The transient was well within the analyzed and anticipated frequency of a turbine trip initiator analyzed in the PRA model and bounded by the Pressure Regulator failure analysis in the USAR.

In conclusion, the safety significance in terms of reactor safety and radiological release to the environment as a result of the transient that occurred on January 10, 2007, was not significant.

Cause

The root cause of this event was determined to be latent shortcomings in the design and construction of the turbine control valve enclosure support system during initial construction. The failure of the support system and associated vertical transition of the control valve enclosure caused the cam that actuates the servo for the control valves to rotate in a direction commanding the control valves to open. This motion overrode signals from the pressure regulating system to close the control valves. Because the cam motion was forced by the transition downward of the control valve enclosure, the control valves remained open despite the turbine trip signal.

Corrective Action

The following actions have been completed:

1. The control valve enclosure support system was redesigned assuring that stress levels remained well below code allowable values. The modified design was installed prior to startup.
2. Collateral damage caused by the failure of the control valve supports and the vertical transition of the enclosure was analyzed and repairs completed as required.
3. Inspections and tests of the turbine control system were performed and repairs affected as necessary to assure full system functionality prior to startup.
4. An extent of condition review and plant walkdown was completed which concluded that other similar support configurations were sufficiently robust such that reasonable assurance of their functionality existed.

Failed Component Identification

None

Previous Similar Events

No previous similar events were identified.